Exelon Generation Company, LLC
Braidwood Station
35100 South Rt 53, Suite 84
Braceville, IL 60407-9619
Tel 815-458-2801

www exeloncorp com

Exelon Nuclear

October 14, 2002
BW020110

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555 - 0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Subject: Pressure and Temperature Limits Reports (PTLRs), Revision 2, Braidwood Station, Units 1 and 2

References: 1) Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, "Pressure and Temperature Limits Report (PTLR), Braidwood Station Unit 1 and Unit 2," dated June 6, 2001

2) Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, "Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report," dated August 30, 2002

The purpose of this letter is to provide recently revised Braidwood Station, Units 1 and 2 Pressure and Temperature Limits Reports (PTLRs). The changes to these PTLRs, identified as Revision 2, do not involve a change in the current NRC approved PTLR methodology and are being submitted in accordance with the reporting requirements of Braidwood Station Technical Specification 5.6.6.c.

Reference 1 provided the May 8, 2001 revision of the Braidwood Station, Units 1 and 2 PTLRs. These PTLRs were revised to accommodate uprated power conditions for Units 1 and 2. As stated in Reference 1, the methods used in this revision of the PTLRs were also consistent with NRC approved PTLR methodology.

During the NRC review of these power uprate PTLRs, several issues were identified that required clarification and correction. Reference 2 provided the response to these NRC issues and committed to issuing corrected PTLRs. Attached are the revised Braidwood PTLRs incorporating the changes requested by the NRC as well as other editorial changes. To assist in the evaluation of the PTLRs, a change list for each Unit's PTLR is also attached.

A001

Please direct any questions you may have regarding this matter to Ms. Amy Ferko, Regulatory Assurance Manager, at (815) 417-2699.

Respectfully,

James D. von Suskil
Site Vice President
Braidwood Station

Attachments:
1. Braidwood Unit 1, Pressure and Temperature Limits Report (PTLR), Revision 2
2. List of Changes to the Unit 1 PTLR
3. Braidwood Unit 2, Pressure and Temperature Limits Report (PTLR), Revision 2
4. List of Changes to the Unit 2 PTLR

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station

Attachment 1

Braidwood Unit 1, Pressure and Temperature Limits Report (PTLR)
Revision 2

BRAIDWOOD UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Revision 2

BRAIDWOOD - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

Table of Contents

List of Figures

List of Tables

BRAIDWOOD - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Introduction

This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 1 has been prepared in accordance with the requirements of Braidwood Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)". Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in this report are listed below:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits; and
LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System.

2.0 Operating Limits

The PTLR limits for Braidwood Unit 1 were developed using a methodology specified in the Technical Specifications. The methodology listed in WCAP-14040-NP-A (Reference 1) was used with the following exceptions:

a) Use of ENDF/B-IV neutron transport cross-section library and ENDF/B-V dosimeter reaction cross-sections,

b) Use of ASME Code Case N-514, and

c) Use of RELAP computer code for calculation of LTOP setpoints for Braidwood Unit 1 replacement steam generators.

These exceptions to the methodology in WCAP 14040-NP-A have been reviewed and accepted by the NRC in Reference 2.

WCAP 14243, Reference 3, provides the basis for the Braidwood Unit 1 P/T curves, along with the best estimate chemical compositions, fluence projections and adjusted reference temperatures used to determine these limits. Reference 4 evaluated the effect of higher fluence from 5% uprate on the existing P/T curves.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3).

2.1.1 The RCS temperature rate-of-change limits defined in Reference 3 are:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, inservice hydrostatic and leak testing, and criticality are specified by Figure 2.1 and Table 2.1a. The RCS P/T limits for cooldown are shown in Figure 2.2 and Table 2.1b. These limits are defined in Reference 3. Consistent with the methodology described in Reference 1 and exceptions noted in Section 2.0, the RCS P/T limits for heatup and cooldown shown in Figures 2.1 and 2.2 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

2.2 Low Temperature Overpressure Protection (LTOP) System Setpoints (LCO 3.4.12).

The power operated relief valves (PORVs) shall each have maximum lift settings in accordance with Figure 2.3 and Table 2.2. These limits are based on References 5, 6, and 7. The Residual Heat Removal (RH) Suction Relief Valves are also analyzed to individually provide low temperature overpressure protection. This analysis for the RH Suction Relief Valves remains valid with the current Appendix G limits contained in this PTLR document and will be reevaluated in the future as the Appendix G limits are revised.

The LTOP setpoints are based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1. The LTOP PORV nominal lift settings shown in Figure 2.3 and Table 2.2 account for appropriate instrument error.

2.3 LTOP Enable Temperature

The minimum required LTOP enable temperature is 200°F (Reference 2).

Braidwood Unit 1 procedures governing the heatup and cooldown of the RCS require the arming of the LTOP System for RCS temperature of 350°F and below and disarming of LTOP for RCS temperature above 350°F.

Note that the last LTOP PORV segment in Table 2.2 extends to 450°F where the pressure setpoint is 2350 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

2.4 Reactor Vessel Boltup Temperature (Non-Technical Specification)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the Reactor Vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

2.5 Reactor Vessel Minimum Pressurization Temperature (Non-Technical Specification)

The minimum temperature at which the Reactor Vessel may be pressurized (i.e., in an unvented condition) shall be ≥ 60°F, plus an allowance for the uncertainty of the temperature instrument, determined using a technique consistent with ISA-S67.04-1994.

MATERIAL PROPERTY BASIS LIMITING MATERIAL: WELD METAL LIMITING ART VALUES AT 14 EFPY: 1/4T, 76 6°F 3/4T, 65 4°F

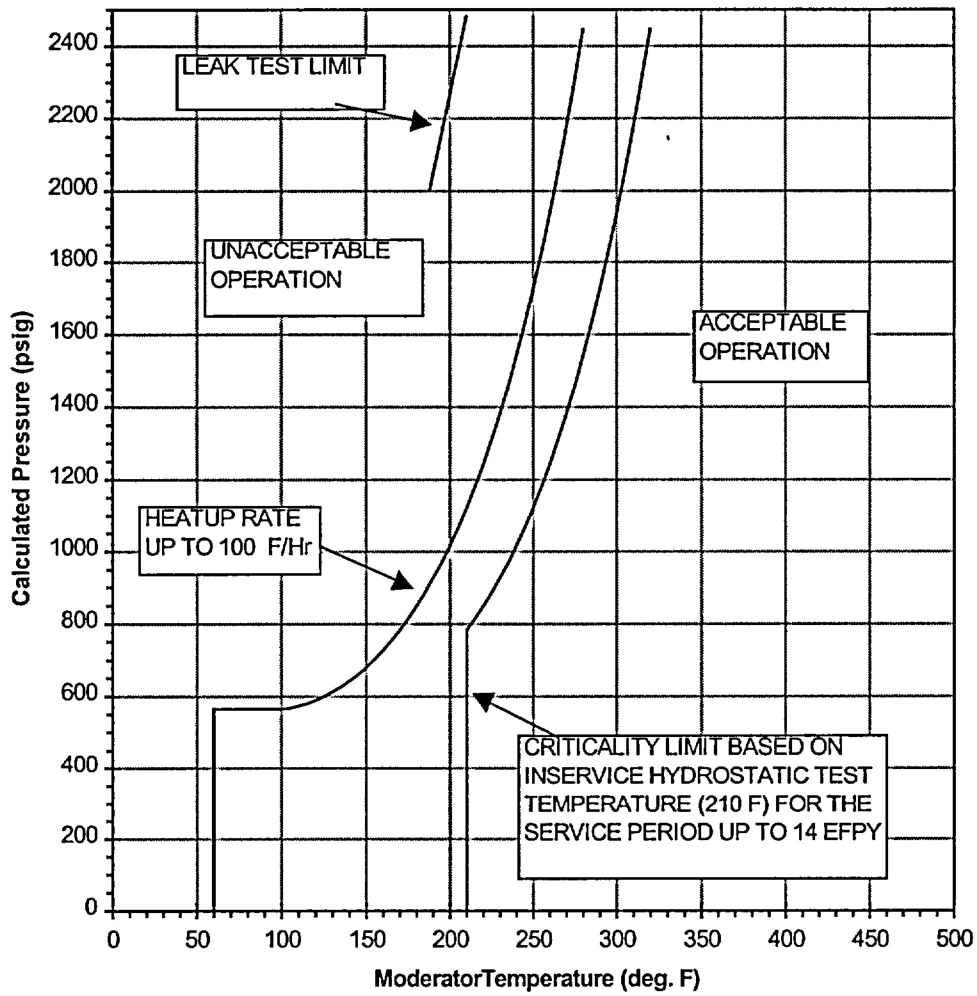

Figure 2.1
Braidwood Unit 1 Reactor Coolant System Heatup Limitations (heatup rate up to 100°F/hr) Applicable for the First 14 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS	
LIMITING MATERIAL WELD METAL	
LIMITING ART VALUES AT 14 EFPY:	1/4T, 76 6°F
	3/4T, 65 4°F

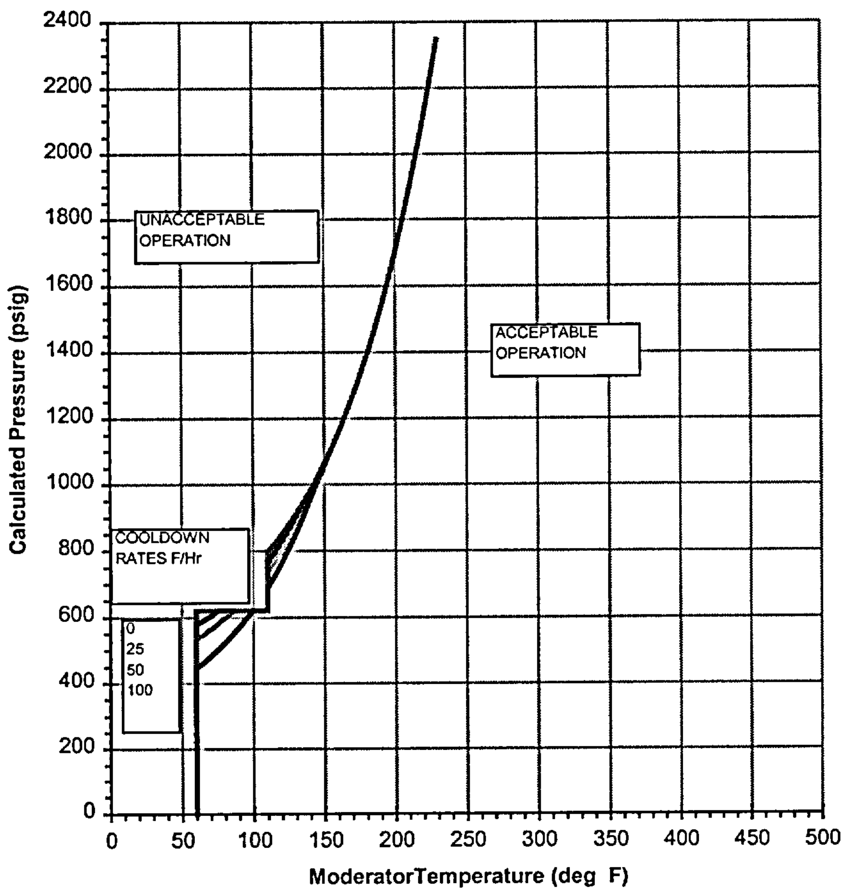

Figure 2.2

Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 0, 25, 50 and 100 °F/hr) Applicable for the First 14 EFPY (Without Margins for Instrumentation Errors)

Table 2.1a
(Page 1 of 2)
Braidwood Unit 1 Heatup* Data Points at 14 EFPY
(Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	210	0	188	2000
60	565.09	210	611.83	210	2485
65	565.09	210	597.56		
70	565.09	210	585.60		
75	565.09	210	576.77		
80	565.09	210	570.35		
85	565.09	210	566.61		
90	565.09	210	565.09		
95	565.09	210	565.87		
100	565.87	210	568.69		
105	568.69	210	573.56		
110	573.56	210	580.30		
115	580.30	210	588.84		
120	588.84	210	599.36		
125	599.36	210	611.78		
130	611.78	210	626.07		
135	626.07	210	642.16		
140	642.16	210	660.36		
145	660.36	210	680.59		
150	680.59	210	702.80		
155	702.80	210	727.33		
160	727.33	210	754.07		
165	754.07	210	783.17		
170	783.17	215	814.98		
175	814.98	220	849.37		
180	849.37	225	886.54		
185	886.54	230	926 73		
190	926.73	235	970.11		
195	970.11	240	1016.91		
200	1016.91	245	1067.33		
205	1067.33	250	1121.63		
210	1121.63	255	1180.01		
215	1180.01	260	1242.62		
220	1242.62	265	1309.84		
225	1309.84	270	1382.03		
230	1382.03	275	1459.45		
235	1459.45	280	1542.27		
240	1542.27	285	1630.97		
245	1630.97	290	1726.05		
250	1726 05	295	1827.80		

Table 2.1a Page 2 of 2					
Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	**P**	**T**	**P**	**T**	**P**
255	1827.80	300	1936.51		
260	1936.51	305	2052.39		
265	2052.39	310	2176.33		
270	2176.33	315	2308.42		
275	2308.42	320	2449.09		
280	2449.09				

* Heatup and Cooldown data includes vessel flange requirements of 110°F and 621 psig per 10CFR50, Appendix G

Table 2.1b
Page 1 of 1
Braidwood Unit 1 Cooldown* Data Points at 14 EFPY**
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	620.27	60	577.45	60	534.28	60	446.98
65	621.00	65	590.68	65	548.52	65	463.79
70	621.00	70	605.03	70	563.98	70	481.93
75	621.00	75	620.51	75	580.67	75	501.49
80	621.00	80	621.00	80	598.51	80	522.68
85	621.00	85	621.00	85	617.90	85	545.50
90	621.00	90	621.00	90	621.00	90	570.23
95	621.00	95	621.00	95	621.00	95	596.83
100	621.00	100	621.00	100	621.00	100	621.00
105	621.00	105	621.00	105	621.00	105	621.00
110	621.00	110	621.00	110	621.00	110	621.00
110	795.92	110	766 92	110	739.27	110	690.04
115	821.55	115	794.59	115	769.53	115	726.24
120	849.00	120	824.45	120	801.97	120	765.12
125	878.42	125	856.54	125	836.87	125	807.07
130	910.25	130	890.97	130	874.41	130	852.23
135	944.34	135	928.00	135	915.03	135	900.91
140	980.89	140	967.79	140	958.57	140	953.33
145	1020.15	145	1010.84	145	1005.42	145	1009.81
150	1062.35	150	1056.88	150	1055.76		
155	1107.92	155	1106.38				
160	1156.42						
165	1208.78						
170	1265.05						
175	1325.37						
180	1390.04						
185	1459.41						
190	1533.55						
195	1613.49						
200	1699.01						
205	1790.55						
210	1888.61						
215	1993.61						
220	2105.69						
225	2225.77						
230	2353.75						

* Heatup and Cooldown data includes vessel flange requirements of 110°F and 621 psig per 10CFR50, Appendix G.

** For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided

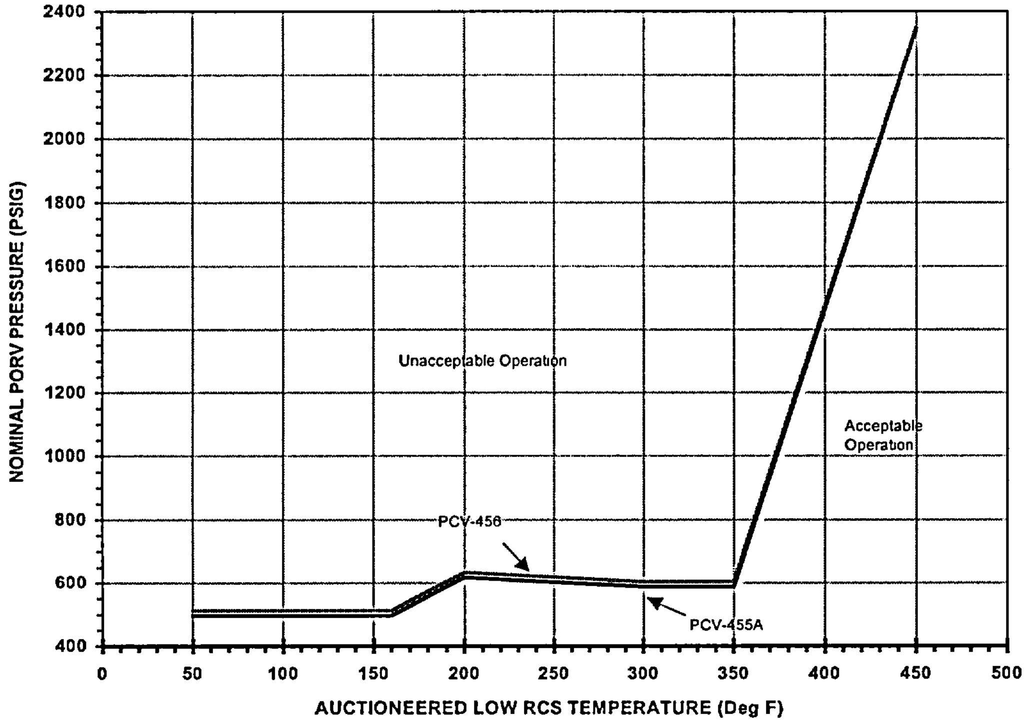

Figure 2.3
Braidwood Unit 1 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the first 14 EFPY

Table 2.2

Data Points for Braidwood Unit 1 Nominal PORV Setpoints for the LTOP System Applicable for the First 14 EFPY

PCV-455A

(1TY-0413M)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	**RCS PRESSURE (PSIG)**
50	497
70	497
100	497
110	497
160	497
200	618
250	603
300	588
350	588
450	2350

PCV-456

(1TY-0413P)	
AUCTIONEERED LOW RCS TEMP. (DEG. F)	**RCS PRESSURE (PSIG)**
50	513
70	513
100	513
110	513
160	513
200	634
250	619
300	604
350	604
450	2350

Note: To determine nominal lift setpoints for RCS Pressure and RCS Temperatures greater than 350°F, linearly interpolate between the 350°F and 450°F data points shown above.

3.0 Reactor Vessel Material Surveillance Program

The pressure vessel material surveillance program (Reference 8) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME Section III, NB-2331. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

The third and final reactor vessel material irradiation surveillance specimens (Capsule W) have been removed and analyzed to determine changes in material properties. The surveillance capsule testing has been completed for the original operating period.

Table 3.1

Braidwood Unit 1 Capsule Withdrawal Schedule

Capsule	Vessel Location (Degrees)	Capsule Lead Factor[a]	Removal Time[b] (EFPY)	Estimated Capsule Fluence (n/cm^2) [a]
U	58.5°	4.37	1.10	3.87×10^{18} [c]
X	238.5°	4.23	4.234	1.24×10^{19} [c]
W	121.5°	4.20	7.61	2.09×10^{19} [c]
Z	301.5°	4.20	Standby	(d)
V	61°	3.92	Standby	(e)
Y	241°	3.92	Standby	(e)

(a) Updated in Capsule W dosimetry analysis, (Reference 9).
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Plant specific evaluation.
(d) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY) Peak Fluence) at approximately 12 EFPY.
(e) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY) Peak Fluence) at approximately 13 EFPY.

4.0 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 4.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data. The values of the CF listed in Table 4.1 are those obtained from the most recent Unit 1 Capsule data, Capsule W, (Reference 9). However, these values were not used in calculating the Adjusted Reference Temperature (ART) values that were used to generate the Braidwood Unit 1 Heatup and Cooldown Curves. The ART values listed in Table 4.3, based on Capsules U and X data, continue to be the basis for the Braidwood Unit 1 curves (Reference 10)

Table 4.2 provides the reactor vessel material properties table.

Table 4.3 provides a summary of the Braidwood Unit 1 adjusted reference temperature (ARTs) at the 1/4T and 3/4T locations for 14 EFPY. The ART values listed in Table 4.3 are based on Capsules U and X data and continue to be the basis for the Braidwood Unit 1 curves (Reference 10).

Table 4.4 shows the calculation of ARTs at 14 EFPY for the limiting Braidwood Unit 1 reactor vessel material, i.e. weld WF-562 (HT # 442011, Based on Surveillance Capsules U and X Data).

Table 4.5 provides RT_{PTS} calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 EFPY), (Reference 11).

Table 4.6 provides RT_{PTS} calculation for Braidwood Unit 1 Beltline Region Materials at Life Extension (48 EFPY), (Reference 11).

TABLE 4.1
Braidwood Unit 1 Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Forging 49D867/49C813-1 (Tangential)	U	0.387	0.737	5.78	4.26	0.543
	X	1.24	1.060	38.23	40.52	1.124
	W	2 09	1.201	24.14	28.99	1.442
Lower Shell Forging 49D867-1 49C813-1 (Axial)	U	0.387	0.737	0.0	0.0	0.543
	X	1.24	1.060	28.75	30.48	1.124
	W	2.09	1.201	37.11	44.57	1.442
				SUM:	148.82	6.218
	$CF_{Forging} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (148.82) \div (6.218) = 23.9°F$					
Braidwood Unit 1 Surv. Weld Material (Heat # 442011)	U	0.387	0.737	17.06	12.57	0.543
	X	1.24	1.060	30.15	31.96	1.124
	W	2.09	1.201	49.68	59.67	1.442
Braidwood Unit 2 Surv. Weld Material (Heat # 442011)	U	0.40	0.746	0.0	0 0	0.557
	X	1.23	1.058	26.3	27.83	1.119
	W	2.25	1.220	23.9	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6\ 273) = 25.7°F$					

Notes:

(a) f = Calculated fluence, (x 10^{19} n/cm², E > 1.0 MeV)

(b) FF = fluence factor = $f^{(0\ 28 - 0\ 1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values.

Table 4.2

Braidwood Unit 1 Reactor Vessel Material Properties

Material Description	Cu (%)	Ni (%)	Chemistry Factor[a]	Initial RT_{NDT} (°F)[a]
Closure Head Flange Heat # 5P7381/3P6406	0.11	0.67		-20
Vessel Flange Heat # 122N357V	--	0.77	--	-10
Nozzle Shell Forging * Heat # 5P-7016	0.04	0.73	26.0°F[b]	10
Intermediate Shell Forging * Heat # 49D383-1/49C344-1 (also referred to as the Upper Shell forging)	0.05	0.73	31.0°F[b]	-30
Lower Shell Forging * Heat # 49D867/49C813-1	0.05	0.74	31.0°F[b] 23.9°F[c]	-20
Circumferential Weld * (Intermediate Shell to Lower Shell) WF-562 (HT# 442011)	0.03	0.67	41.0°F[b] 25.7°F[c]	40
Upper Circumferential Weld * (Nozzle Shell to Intermediate Shell) WF-645 (HT# H4498)	0.04	0.46	54.0°F[b]	-25

* Beltline Region Materials

a) The Initial RT_{NDT} values for the plates and welds are based on measured data.

b) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 1.1.

c) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 2 1.

Table 4.3

Summary of Braidwood Unit 1 Adjusted Reference Temperatures (ARTs) at 1/4T and 3/4T Locations for 14 EFPY[c]

Material Description	14 EFPY	
	1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Forging Heat # 49D383-1/49C344-1 (RG Position 1)	25.1	8.2
Lower Shell Forging Heat # 49D867/49C813-1 (RG Position 1)	26.2	12.1
Using Surveillance Data[a] (RG Position 2[a])	13.4	3.2
Circumferential Weld (Intermediate Shell to Lower Shell) WF-562 (HT# 442011) (RG Position 1)	112.9	90.5
Using credible surveillance Data (RG Position 2[a])	76.6[b]	65.4[b]

(a) Calculated using a chemistry factor based on Regulatory Guide (RG) 1.99, Position 2.
(b) These ART values were used to generate the Braidwood Unit 1 Heatup and Cooldown curves, (Reference 3).
(c) The applicability date has been decreased to 14 EFPY from 16 EFPY to reflect the updated chemistry and uprated fluence values (Reference 10).

Table 4.4		
Braidwood Unit 1 Calculation of Adjusted Reference Temperatures (ARTs) at 14 EFPY[b] at the Limiting Reactor Vessel Material Weld Metal (Based on Surveillance Capsule Data)		
Parameter	Values	
Operating Time	14 EFPY[b]	
Location[c]	1/4T ART(°F)	3/4T ART(°F)
Chemistry Factor, CF (°F)	20.6	20.6
Fluence(f), n/cm^2 (E>1.0 Mev)[a]	6.73×10^{18}	2.43×10^{18}
Fluence Factor, FF	0.889	0.616
ΔRT_{NDT}= CFxFF(°F)	18.31	12.70
Initial $RT_{NDT,}$, I(°F)	40	40
Margin, M (°F)	18.31	12.70
ART= I+(CF*FF)+M,°F per RG 1.99, Revision 2	76.6	65.4

(a) Fluence f, is based upon f_{surf} (E > 1.0 Mev) = 1.120×10^{19} at 14 EFPY for uprated conditions.
(b) The applicability date has been decreased to 14 EFPY from 16 EFPY to reflect the updated chemistry and uprated fluence values.
(c) The Braidwood Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.

Table 4.5

RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence (10^{19} n/cm², E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging Heat # 49D383-1/49C344-1	2.05	1.20	31.0	37.2	34	-30	41
Lower Shell Forging Heat # 49D867/49C813-1	2.05	1.20	31.0	37.2	34	-20	51
Lower Shell Forging (Using S/C Data)	2.05	1.20	23.9	28.7	17	-20	26
Nozzle Shell Forging Heat # 5P-7016	0.608	0.86	26.0	22.4	22.4	10	55
Inter. to Lower Shell Circ. Weld WF-562 (HT# 442011)	1.99	1.19	41.0	48.8	48.8	40	138
Inter. to Lower Shell Circ. Weld Using S/C Data	1.99	1.19	25 7	30.6	28	40	99
Nozzle Shell to Inter. Shell Circ. Weld WF-645 (HT# H4498)	0.608	0.86	54.0	46.5	46.5	-25	68

(a) Initial RT_{NDT} values are measured values.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(c) $\Delta RT_{PTS} = CF * FF$

Table 4.6

RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence[a] (10^{19}n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Forging Heat # 49D383-1/49C344-1	3.06	1.30	31.0	40.3	34	-30	44
Lower Shell Forging Heat # 49D867/49C813-1	3.06	1.30	31.0	40.3	34	-20	54
Lower Shell Forging Using S/C Data	3.06	1.30	23.9	31.1	31.1	-20	42
Nozzle Shell Forging Heat # 5P-7016	0.909	0.97	26.0	25.2	25.2	10	60
Inter. to Lower Shell Circ. Weld Metal WF-562 (HT# 442011)	2.98	1.29	41.0	52.9	52.9	40	146
Inter. to Lower Shell Circ. Weld Using S/C Data	2.98	1.29	25.7	33.2	28	40	101
Nozzle Shell to Inter. Shell Circ. Weld Metal WF-645 (HT# H4498)	0.909	0.97	54.0	52.4	52.4	-25	80

(a) Initial RT_{NDT} values are measured values.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(c) ΔRT_{PTS} = CF * FF

5.0 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Andrachek, J.D., et. al., January 1996.

2. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.

3. WCAP-14243, "Commonwealth Edison Company, Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," March 1995.

4. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY."

5. Westinghouse Letter to Commonwealth Edison Company, CCE-95-186, "Braidwood Unit 1 LTOPS Setpoints Based on 16 EFPY P/T Limits," June 5, 1995.

6. ComEd Calculation BRW-96-906I/BYR 96-293, "Channel Accuracy for Power Operated Reief Valve (PORV) Setpoints and Wide Range RCS Temperature Indication (Unit 1 Original Steam Generators and Replacement Steam Generators)," Revision 0.

7. ComEd Nuclear Fuel Services Department, NDIT No. 960194, "Maximum Allowable LTOPS PORV Setpoints for Braidwood Unit 1 with RSGs," Revision 2.

8. WCAP-9807, "Commonwealth Edison Company, Braidwood Station Unit 1 Reactor Vessel Radiation Surveillance Program," February 1981.

9. WCAP-15316, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," December 1999.

10. Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, "Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report", dated August 30, 2002.

11. WCAP-15365, Revision 1, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1," September 2000.

Attachment 2

List of Changes to the Unit 1 PTLR

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
ii and iii	List of Figures/ Tables	The description of Figures 2.1 and 2..2 and Tables 2.1a and 2.1b describe the heatup rate curve as "using 1996 Appendix G Methodology."	This description is removed.	The Unit 1 PTLR does not use ASME Section XI 1995 edition, 1996 addenda. The Unit 1 PTLR curves are generated using Appendix G of the 1989 Edition of the ASME Section XI Code. The Unit 1 PTLR uses ASME Code Case N-514, "Low Temperature Overpressure Protection," which was later adopted into the 1995 edition with 1996 addenda. See WCAP-14243, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operations," Section 6.0.
1	1.0	"This PTLR for Unit 1 has been prepared in accordance with the requirements of TS 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance."	"This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 1 has been prepared in accordance with the requirements of Braidwood Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)". Revisions to the PTLR shall be provided to the NRC after issuance."	Editorial change. Sentence reworded for clarification.
1	2.0	Paragraph a lists an exception from the approved WCAP-14040-NP-A methodology as: "a) Optional Use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda	This exception is removed and the exceptions, with modification, listed in the pre-power uprate PTLR are restored. "a) Use of ENDF/B-IV neutron transport cross-section library and ENDF/B-V dosimeter reaction cross-sections, b) Use of ASME Code case N-514,	Established and approved in pre-Power Uprate PTLR for Unit 1, a) R. A. Capra to O. D. Kingsley letter, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated January 21, 1998, see SER page 2, section 2.1 b) R. A. Assa to D. L. Farrar letter, "Exemption from Requirements of 10 CFR 50.60 - Braidwood Station Unit 1," dated July 13, 1995 c) R. A. Capra to O. D. Kingsley letter,

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
1	2.0	b) Use of RELAP computer code for calculation of LTOP setpoints for Braidwood Unit 1 replacement steam generators."	c) Use of RELAP computer code for calculation of LTOP setpoints for Braidwood Unit 1 replacement steam generators."	"Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated January 21, 1998, see SER page 10.
1	2.0	"Reference 17"	Revised to "Reference 2." Note - Reference numbers throughout the Unit 1 PTLR have be changed to reflect either changes in the actual reference or changes in the numbering of existing references.	The basis for actual reference changes will be addressed in the Reference section of this change list. The renumbering is an editorial change.
2	2.1.2	Paragraph contains the statement:: "These limits were developed using ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda.	This statement has been deleted.	The Unit 1 PTLR does not use ASME Section XI 1995 edition, 1996 addenda. The Unit 1 PTLR curves are generated using Appendix G of the 1989 Edition of the ASME Section XI Code. The Unit 1 PTLR uses ASME Code Case N-514, "Low Temperature Overpressure Protection," which was later adopted into the 1996 edition with 1996 addenda. See WCAP-14243, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operations," Section 6.0.
2	2.3	"The required enable temperature for the PORVs shall be [greater than or equal to] 350 degrees F RCS temperature."	Deleted. The statement is confusing.	The remaining statements on the Braidwood Procedures for arming and disarming the LTOP are sufficient.
3	2.4	"Reference 7"	Deleted. There is not a reference for this statement.	Editorial
12	Table 3.1 footnote "a"	"Updated in Capsule W dosimetry analysis."	"Updated in Capsule W dosimetry analysis (Reference 9)."	Added reference to WCAP-15316, " Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," December 1999 for clarification.

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
13	4.0	"Table 4.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data."	Added an additional sentence - "The values of the CF listed in Table 4.1 are those obtained from the most recent Unit 1 Capsule data, Capsule W, (Reference 9). However, these values were not used in calculating the Adjusted Reference Temperature (ART) values that were used to generate the Braidwood Unit 1 Heatup and Cooldown Curves. The ART values listed in Table 4.3, based on Capsules U and X data, continue to be the basis for the Braidwood Unit 1 curves (Reference 10)."	Words added to clarify the use of Capsule W data. Reference 10 is new. Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report , dated August 30, 2002
13	4.0	"Table 4.3 provides a summary of the Braidwood Unit 1 adjusted reference temperature (ARTs) at the 1/4T and 3/4T locations for 14 EFPY."	Added an additional sentence - " The ART values listed in Table 4.3 are based on Capsules U and X data and continue to be the basis for the Braidwood Unit 1 curves (Reference 10)."	Words added to clarify the use of Capsule W data. Reference 10 is new. Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report , dated August 30, 2002
13	4.0	"Table 4.4 shows the calculation of ARTs at 14 EFPY for the limiting Braidwood Unit 1 reactor vessel material, i.e. weld metal HT # 442011, (Based on Surveillance Capsule Data)."	Added weld number - weld WF-562 (HT # 442011) and additional wording within parentheses - "(Based on Surveillance Capsules U and X Data)"	Words added to clarify the use of Capsule W data. Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report , dated August 30, 2002
13	4.0	Listing for Table 4.5 has no reference	Added "(Reference 11)"	Editorial, WCAP-15365 Rev 1 provides the basis for this Table.
13	4.0	Listing for Table 4.6 has no reference	Added "(Reference 11)"	Editorial, WCAP-15365 Rev 1 provides the basis for this Table.

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
15	Table 4.2	Clarifications and additions to Table 4.2 listings	1) For Closure Head Flange, added "Heat #" and corrected heat to 5P7381/3P6406 2) For Vessel Flange, added "Heat #" and corrected heat by dropping "A1" from end of number. 2a) For Nozzle Shell Forging, added "Heat #" 3) For Inter. Shell Forging, spelled out intermediate, added "Heat #" and added note - "(also referred to as the Upper Shell forging)" to provide cross reference to NRC RVID. 4) For Lower Shell Forging, added "Heat #" 5) For Circumferential Weld, added clarifying note - "(Intermediate Shell to Lower Shell)" 6) For Upper Circumferential Weld, added clarifying note - "(Nozzle Shell to Intermediate Shell)" 7) Added Note "Beltline Region Materials" to apply to Nozzle Shell Forging, Intermediate Shell Forging, Lower Shell Forging, and the two welds. ***Note Heat Numbers were also added to the materials listed in Tables 4.3, 4.5, and 4.6 ***	1) WCAP-15205, "Compilation of Available Westinghouse Information for the Materials of the Braidwood Unit 1 Reactor Vessel" 2) WCAP-15205 2a) Editorial 3) Editorial and NRC RVID 2 Database, D. Saccomando to U. S. NRC letter, "ComEd Response to NRC Generic Letter 92-01,' Rev1, Supplement 1: Reactor Vessel Structural Integrity," dated November17, 1995.. 4) Editorial 5) Editorial, WCAP-15366, Rev 1, " Braidwood Unit 1 Surveillance Program Credibility Evaluation" 6) Editorial, WCAP-15366, Rev 1 7) Clarification of materials of interest, WCAP-15366, Rev 1 Editorial, for clarification.

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
16	Table 4.3	Material Descriptions contained Construction Piece (mark) Numbers	Revised to use heat numbers rather than mark numbers Intermediate Shell Forging 24-2 became Intermediate Shell Forging Heat # 49D383-1/49C344-1 Lower Shell Forging 24-3 became Lower Shell Forging Heat # 49D867/49C813-1 Circumferential Weld became Circumferential Weld (Intermediate Shell to Lower Shell) WF-562 (HT# 442011)	Editorial, provides consistency between Table 4.2 and 4.3.
16	Table 4.3	Footnote (a) lists Reg Guide 1.99 as Reference 11	Delete listing of Reference 11	Editorial, the reference is self contained in the footnote.
16	Table 4.3	Footnote (b) lists WCAP-14243 as Reference 7.	Revise to Reference 3 and delete " WCAP-14243"	Editorial, no need to repeat reference wording.
16	Table 4.3	Footnote (c) has no reference	Add (Reference 10)	Editorial, identifies the basis for the note.
18	Table 4.5	The CF, Delta RT PTS, Margin, and RT PTs values in the Table for the Nozzle Shell to Inter. Shell Circ Weld Metal are incorrect.	Revised - CF from 46.0 to 54.0 Delta RT PTS from 39.6 to 46.5 Margin from 39.6 to 46.5 RT PTS from 54 to 68	Identified during NRC review. WCAP-15365, Rev 1, Table 5 0
19	Table 4.6	The CF, Delta RT PTS, Margin, and RT PTs values in the Table for the Nozzle Shell to Inter. Shell Circ Weld Metal are incorrect.	Revised - CF from 46.0 to 54.0 Delta RT PTS from 44 6 to 52.4 Margin from 44.6 to 52.4 RT PTS from 64 to 80.	Identified during NRC review. WCAP-15365, Rev 1, Table 6

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
20	5.0	1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Andrachek, J.D., et. al., January 1996.	Maintained as Reference 1.	The retained references were renumbered to be sequential in the PTLR. Previous References 2, 3, 4, 9, 10, 11, 12, 13, 14, 15 , and 18 were deleted. These references were either not used in support of the Unit 1 PTLR or were not required to be specifically identified as a reference, (e.g. Ref.11 is specifically identified within the PTLR).
		2. Laubham, T.A. et al., WCAP-14824, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration for Byron and Braidwood", Revision 2, November 1997 and Errata Sheets (Westinghouse Letter CAE-97-220, CCE-97-304 and Westinghouse Letter CAE-97-233, CCE-97-316).	Deleted, replaced by: 2. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.	Old reference 2 was not used.
		3. WCAP-14241, "Analysis of Capsule X from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," March 1995.	Deleted, replaced by: 3. WCAP-14243, "Commonwealth Edison Company, Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," March 1995.	Old reference 3 was not used.
		4. WCAP – 12685, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," August 1990.	Deleted, replaced by: 4. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY"	Old reference 4 was not used.
		5. Westinghouse Letter to Commonwealth Edison Company, CCE-95-186, "Braidwood Unit 1 LTOPS Setpoints Based on 16 EFPY P/T Limits," June 5, 1995.	Maintained as Reference 5	

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
		6. WCAP-9807, "Commonwealth Edison Company, Braidwood Station Unit 1 Reactor Vessel Radiation Surveillance Program," Yanichko, S.E., et al., February 1981..	Became Reference 8). Reference 6 is now : 6)ComEd Calculation BRW-96-906I/BYR 96-293, "Channel Accuracy for Power Operated Relief Valve (PORV) Setpoints and Wide Range RCS Temperature Indication (Unit 1 Original Steam Generators and Replacement Steam Generators)," Revision 0.	Editorial. Revised References 6 and 7 were added (actually, restored from the pre-Power Uprate PTLR) since they are still the documents which provided the basis for the Unit 1 LTOP Setpoints.
		7. WCAP-14243, "Commonwealth Edison Company, Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," March 1995.	Became Reference 3). Reference 7 is now: 7. ComEd Nuclear Fuel Services Department, NDIT No. 960194, "Maximum Allowable LTOPS PORV Setpoints for Braidwood Unit 1 with RSGs," Revision 2.	Editorial.
		8. WCAP-15365, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1," September 2000, Terek, E.	Became Reference 11. Reference 8 is now. 8. WCAP-9807, "Commonwealth Edison Company, Braidwood Station Unit 1 Reactor Vessel Radiation Surveillance Program," February 1981.	Editorial.
		9. NOT USED	Deleted. Reference 9 is now: 9. WCAP-15316, " Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," December 1999	
		10. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," (PTS Rule) January 18, 1996.	Deleted. Reference 10 is now:	Revised Reference 10 was added to address the changes required from the recent NRC RAI.

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
			10. Letter from J. D. von Suskil (Exelon Generation Company, LLC) to U.S. NRC, Braidwood Station Response to U. S. NRC Request for Additional Information Regarding the Braidwood Station Pressure-Temperature Limits Report , dated August 30, 2002	
		11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.	Deleted. Reference 11 is now: 11. WCAP-15365, Revision 1, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1," September 2000.	Revised Reference 11 was added since it provides the basis for the PTS evaluations in Tables 4.5 and 4.6. Old reference 11 was not required.
		12. WCAP-14970, "Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation, " October 1997 and Errata Sheets (Westinghouse Letter CAE-97-210, CCE-97-289 and Westinghouse Letter CAE-97-232 and CCE-97-315).	Deleted.	Old reference 12 was not used.
		13. Exelon Document ID # DG01-000125, "Power Uprate-Unit 2 LTOPS," R.D. Koenig, dated February 20, 2001.	Deleted.	Old reference 13 was not used.
		14. CAE-00-164, "Cold Overpressure Mitigation System Setpoint Analysis for Braidwood Units 1 and 2 Uprating Program", dated June 19, 2000.	Deleted.	Old reference 14 was not used.
		15. WCAP-15626, "Braidwood Unit 2 12 and 14 EFPY Heatup and Cooldown Limit Curves for Normal Operation Using Uprated Fluences", J.H. Ledger, January 2000.	Deleted.	Old reference 15 was not used.

Attachment 2
List of Changes to the Braidwood Unit 1 PTLR

Page	Section	Previous	Revised to	Basis
		16. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY"	Became Reference 4	Editorial.
		17. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.	Became Reference 2.	Editorial.
		18. CAE-01-016, "Exelon Nuclear Byron and Braidwood Units 1 and 2 Power Uprate Project Additional Information for Byron Units 1 and 2 and Braidwood Unit 1 P/T Curve Information", dated February 8,2001.	Deleted.	Old reference 18 was not used.
		19. WCAP-15316, " Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program," December 1999.	Became Reference 9.	Editorial.

Attachment 3

Braidwood Unit 2, Pressure and Temperature Limits Report (PTLR), Revision 2

BRAIDWOOD UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

Revision 2

BRAIDWOOD - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

Table of Contents

List of Figures

List of Tables

BRAIDWOOD - UNIT 2
PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Introduction

This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 2 has been prepared in accordance with the requirements of Braidwood Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)". Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits; and
LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System.

2.0 Operating Limits

The PTLR limits for Braidwood Unit 2 were developed using a methodology specified in the Technical Specifications. The methodology listed in WCAP-14040-NP-A (Reference 1) was used with the following exception:

a) Optional use of ASME Code Section XI, Appendix G, Article G-2000, 1996 Addenda,

This exception to the methodology in WCAP 14040-NP-A has been reviewed and accepted by the NRC in Reference 2.

WCAP 15626, Reference 3, provides the basis for the Braidwood Unit 2 P/T curves, along with the best estimate chemical compositions, fluence projections and adjusted reference temperatures used to determine these limits. Reference 4 evaluated the effect of higher fluence from 5% uprate on the existing P/T curves.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3).

2.1.1 The RCS temperature rate-of-change limits defined in Reference 3 are:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, inservice hydrostatic and leak testing, and criticality are specified by Figure 2.1 and Table 2.1a. The RCS P/T limits for cooldown are shown in Figure 2.2 and Table 2.1b. These limits are defined in

Reference 3. Consistent with the methodology described in Reference 1, with the exception noted in Section 2.0, the RCS P/T limits for heatup and cooldown shown in Figures 2.1 and 2.2 are provided without margins for instrument error. These limits were developed using ASME Code Section XI, Appendix G, Article G2000, 1996 Addenda. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

2.2 Low Temperature Overpressure Protection (LTOP) System Setpoints (LCO 3.4.12).

The power operated relief valves (PORVs) shall each have nominal lift settings in accordance with Figure 2.3 and Table 2.2. These limits are based on Reference 5. The Residual Heat Removal (RH) Suction Relief Valves are also analyzed to individually provide low temperature overpressure protection. This analysis for the RH Suction Relief Valves remains valid with the current Appendix G limits contained in this PTLR document and will be reevaluated in the future as the Appendix G limits are revised.

The LTOP setpoints are based on P/T limits that were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error. The LTOP setpoints were developed using the methodology described in Reference 1. The LTOP PORV nominal lift settings shown in Figure 2.3 and Table 2.2 account for appropriate instrument error.

2.3 LTOP Enable Temperature

The minimum required LTOP enable temperature is 200°F (Reference 6).

Braidwood Unit 2 procedures governing the heatup and cooldown of the RCS require the arming of the LTOP System for RCS temperature of 350°F and below and disarming of LTOP for RCS temperature above 350°F.

Note that the last LTOP PORV segment in Table 2.2 extends to 450°F where the pressure setpoint is 2335 psig. This is intended to prohibit PORV lift for an inadvertent LTOP system arming at power.

2.4 Reactor Vessel Boltup Temperature (Non-Technical Specification)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the Reactor Vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

2.5 Reactor Vessel Minimum Pressurization Temperature (Non-Technical Specification)

The minimum temperature at which the Reactor Vessel may be pressurized (i.e., in an unvented condition) shall be ≥ 60°F, plus an allowance for the uncertainty of the temperature instrument, determined using a technique consistent with ISA-S67.04-1994.

Material Property Basis
Limiting Material: Weld Metal
Limiting ART Values at 14 EFPY 1/4T 82°F
3/4T 68°F

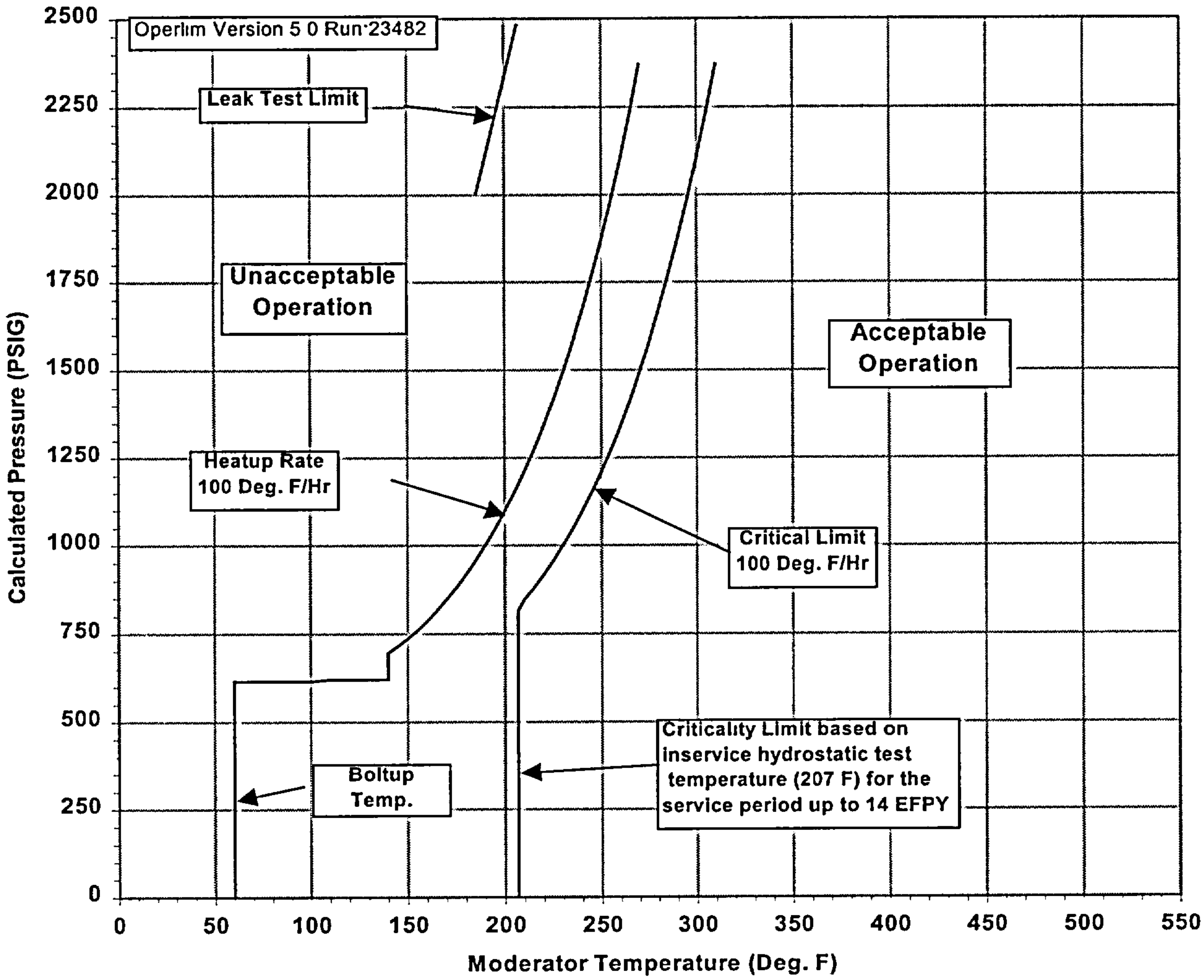

Figure 2.1
Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 14 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)

Material Property Basis
Limiting Material: Weld Metal
Limiting ART Values at 14 EFPY 1/4T 82°F
3/4T 68°F

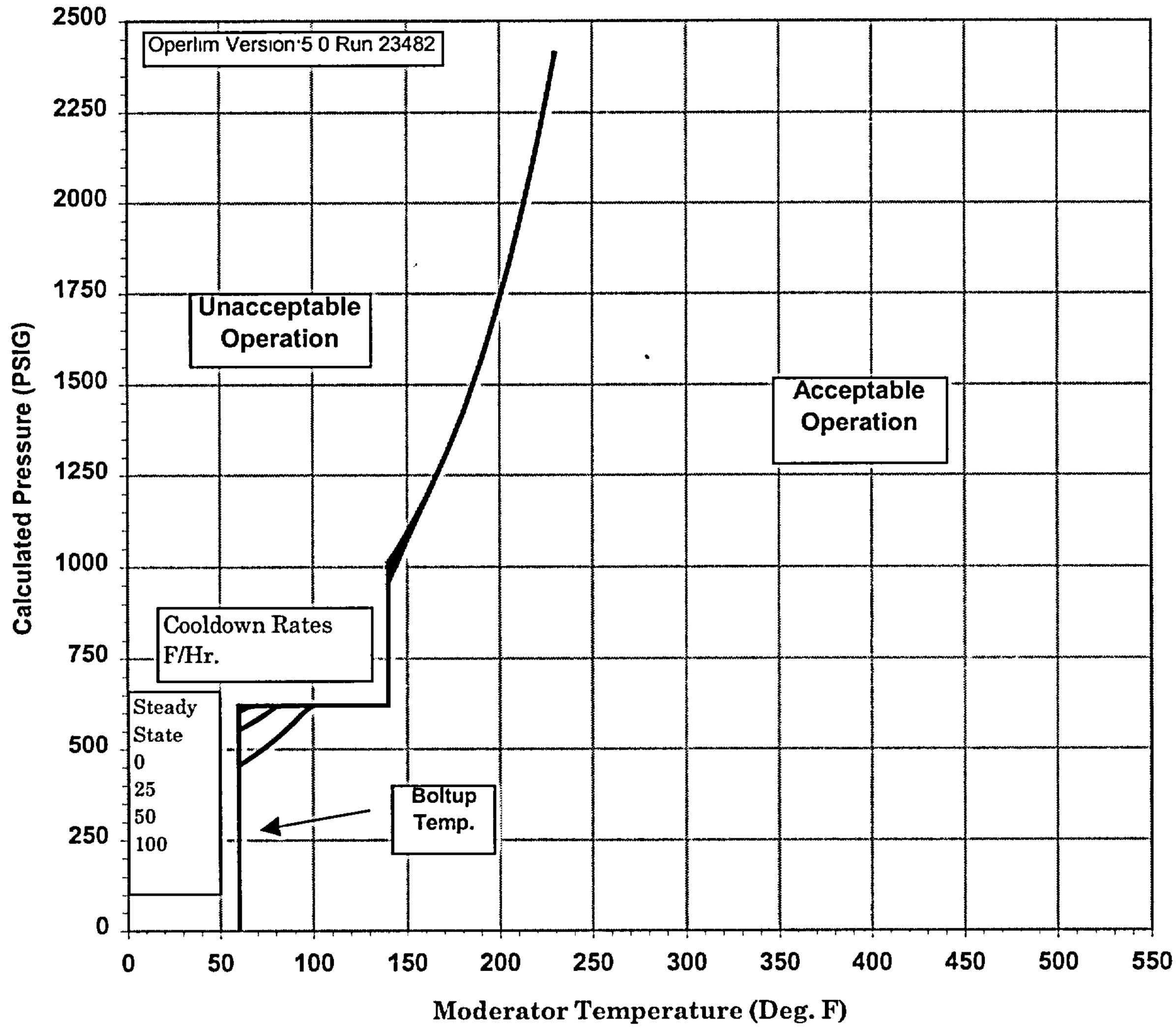

Figure 2.2
Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to the First 14 EFPY using 1996 Appendix G Methodology (Without Margins of Instrumentation Errors)

Table 2.1a
(Page 1 of 2)

Braidwood Unit 2 Heatup* Data Points at 14 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)

Heatup Curve					
100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	207	0	186	2000
60	617	207	621	207	2485
65	617	207	621		
70	617	207	621		
75	617	207	621		
80	617	207	621		
85	617	207	621		
90	617	207	621		
95	617	207	621		
100	617	207	621		
105	619	207	621		
110	621	207	621		
115	621	207	621		
120	621	207	621		
125	621	207	621		
130	621	207	621		
135	621	207	621		
140	621	207	696		
140	621	207	715		
140	696	207	736		
145	715	207	760		
150	736	207	786		
155	760	207	815		
160	786	210	846		
165	815	215	880		
170	846	220	917		
175	880	225	957		
180	917	230	1000		
185	957	235	1047		
190	1000	240	1097		
195	1047	245	1152		
200	1097	250	1210		
205	1152	255	1273		

Table 2.1a
Page 2 of 2

Heatup Curve

100 F Heatup		Criticality Limit		Leak Test Limit	
T	P	T	P	T	P
210	1210	260	1341		
215	1273	265	1415		
220	1341	270	1493		
225	1415	275	1578		
230	1493	280	1669		
235	1578	285	1766		
240	1669	290	1871		
245	1766	295	1984		
250	1871	300	2105		
255	1984	305	2235		
260	2105	310	2374		
265	2235				
270	2374				

* Heatup and Cooldown data includes the vessel flange requirements of 140 °F and 621 psig per 10CFR50, Appendix G

Table 2.1b
(Page 1 of 1)

Braidwood Unit 2 Cooldown* Data at 14 EFPY Using the 1996 Appendix G Methodology (Without Margins for Instrumentation Errors)**

Cooldown Curves							
Steady State		25 °F Cooldown		50 °F Cooldown		100 °F Cooldown	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	621	60	602	60	554	60	455
65	621	65	616	65	568	65	471
70	621	70	621	70	583	70	489
75	621	75	621	75	599	75	508
80	621	80	621	80	617	80	529
85	621	85	621	85	621	85	552
90	621	90	621	90	621	90	576
95	621	95	621	95	621	95	603
100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621
140	621	140	621	140	621	140	621
140	1010	140	991	140	975	140	957
145	1050	145	1034	145	1022	145	1013
150	1092	150	1080	150	1072	150	1074
155	1137	155	1129	155	1126	155	1137
160	1186	160	1183	160	1185	160	1186
165	1239	165	1239	165	1239	165	1239
170	1295	170	1295	170	1295	170	1295
175	1356	175	1356	175	1356	175	1356
180	1422	180	1422	180	1422	180	1422
185	1492	185	1492	185	1492	185	1492
190	1567	190	1567	190	1567	190	1567
195	1649	195	1649	195	1649	195	1649
200	1736	200	1736	200	1736	200	1736
205	1830	205	1830	205	1830	205	1830
210	1931	210	1931	210	1931	210	1931
215	2039	215	2039	215	2039	215	2039
220	2156	220	2156	220	2156	220	2156
225	2281	225	2281	225	2281	225	2281
230	2416	230	2416	230	2416	230	2416

* Heatup and Cooldown data includes the vessel flange requirements of 140 °F and 621 psig per 10CFR50, Appendix G

** For each cooldown rate, the steady-state pressure values shall govern the temperature where no allowable pressure values are provided

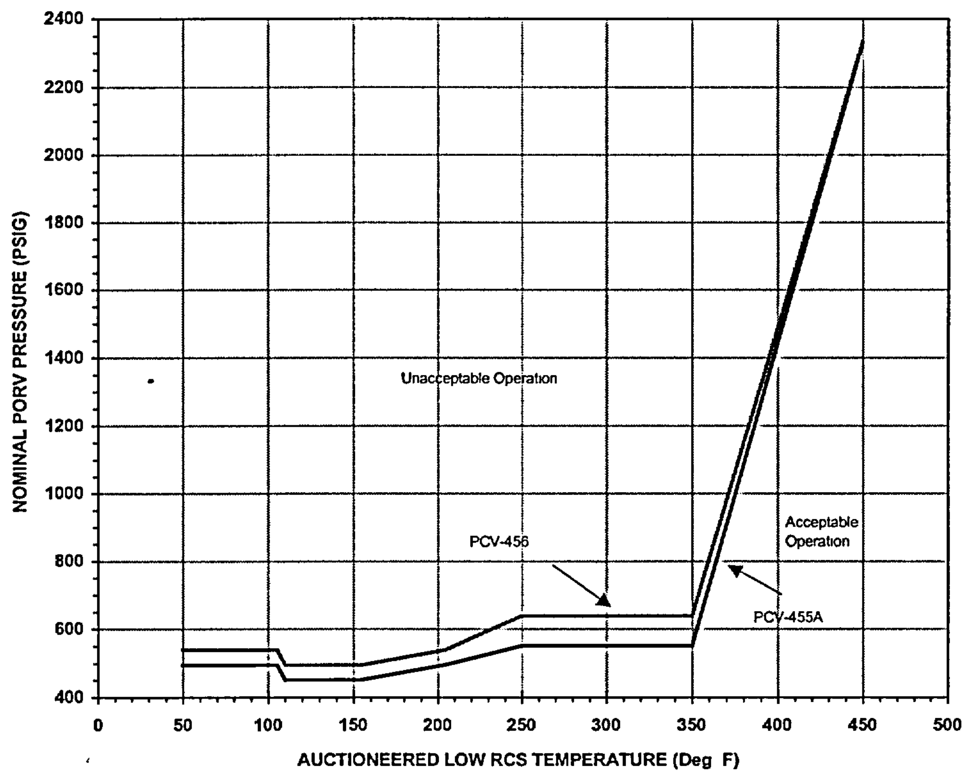

Figure 2.3
Braidwood Unit 2 Nominal PORV Setpoints for the Low Temperature Overpressure Protection (LTOP) System Applicable for the First 14 EFPY

Table 2.2

Data Points for Braidwood Unit 2 Nominal PORV Setpoints for the LTOP System Applicable for the First 14 EFPY

PCV-455A

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
50	495.8
105	495.8
110	451.0
155	451.0
205	496.4
250	551.7
350	551.7
450	2335.0

PCV-456

RCS TEMP. (DEG. F)	RCS Pressure (PSIG)
50	539.5
105	539.5
110	496.0
155	496.0
205	540.1
250	639.0
350	639.0
450	2335.0

Note: To determine nominal lift setpoints for RCS Pressure and RCS Temperatures greater than 350°F, linearly interpolate between the 350°F and 450°F data points shown above. (Setpoints extend to 450°F to prevent PORV liftoff from an inadvertent LTOP system arming while at power).

3.0 Reactor Vessel Material Surveillance Program

The pressure vessel material surveillance program (Reference 7) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME, Section III, NB-2331. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

The third and final reactor vessel material irradiation surveillance specimens (Capsule W) have been removed and analyzed to determine changes in material properties. The surveillance capsule testing has been completed for the original operating period.

Table 3.1

Braidwood Unit 2 Capsule Withdrawal Schedule

Capsule	Location (Degrees)	Capsule Lead Factor[a]	Removal Time[b] (EFPY)	Estimated Capsule Fluence (n/cm^2) [a]
U	58.5°	4.41	1.15	4.00×10^{18} (c)
X	238.5°	3.85	4.215	1.23×10^{19}(c)
W	121.5°	4.17	8.53	2.25×10^{19}(c)
Z	301.5°	4.17	Standby	(d)
V	61.0°	3.92	Standby	(e)
Y	241.0°	3.92	Standby	(e)

Notes:
(a) Updated in Capsule W dosimetry analysis (Reference 8)
(b) Effective Full Power Years (EFPY) from plant startup.
(c) Plant specific evaluation.
(d) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 12 EFPY.
(e) This capsule will reach a fluence of approximately 2.94×10^{19} (48 EFPY Peak Fluence) at approximately 13 EFPY.

4.0 Supplemental Data Table

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 4.1 shows the calculation of the surveillance material chemistry factors using surveillance capsule data (Reference 8).

Table 4.2 provides the reactor vessel material properties table.

Table 4.3 provides a summary of the Braidwood Unit 2 adjusted reference temperatures (ARTs) at the 1/4T and 3/4T locations for 14 EFPY.

Table 4.4 shows the calculation of ARTs at 14 EFPY for the limiting Braidwood Unit 2 reactor vessel material.

Table 4.5 provides RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY), (Reference 9).

Table 4.6 provides RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at Life Extension (48 EFPY), (Reference 9).

Table 4.1

Braidwood Unit 2 Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT} [c]	FF*ΔRT_{NDT}	$(FF)^2$
Lower Shell Forging (50D102-1/50C97-1) (Tangential)	U	0.400	0.746	0.0	0.0	0.557
	X	1.23	1.058	0.0	0.0	1.119
	W	2.25	1.220	4.53	5.53	1.488
Lower Shell Forging (50D102-1/50C97-1) (Axial)	U	0.400	0.746	0.0	0.0	0.557
	X	1.23	1.058	33.94	35.91	1.119
	W	2.25	1.220	33.2	40.50	1.488
				Sum:	81.94	6.328
	Chemistry Factor = $\Sigma(FF*\Delta RT_{NDT}) \div \Sigma(FF^2) = (81.94) \div (6.328) = 12.9°F$					
Braidwood 1 Surv.Weld Material	U	0.387	0.737	17.06 [d]	12.57	0.543
	X	1.24	1.060	30.15 [d]	31.96	1.124
	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood 2 Surv. Weld Material	U	0.40	0.746	0.0	0.0	0.557
	X	1.23	1.058	26.3[d]	27.83	1.119
	W	2.25	1.220	23.9[d]	29.16	1.488
				Sum:	161.19	6.273
	Chemistry Factor = $\Sigma(FF*\Delta RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

NOTES:

(a) f = Calculated fluence, (x 10^{19} n/cm^2, E > 1.0 MeV)

(b) FF= fluence factor = $f^{(0.28 - 0.1 * \log f)}$

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted.

Table 4.2
Braidwood Unit 2 Reactor Vessel Material Properties

Material Description	Cu (%)	Ni (%)	Chemistry Factor[a]	Initial RT_{NDT} (°F) [a]
Closure Head Flange Heat # 3P6566/5P7547/4P6986 Serial # 2031-V-1	---	0.75		20
Vessel Flange Heat # 124P455	0.07	0.70		20
Nozzle Shell Forging * Heat # 5P7056	0.04	0.90	26.0°F[b]	30
Intermediate Shell Forging * Heat # 49D963/49C904-1-1) (also referred to as the Upper Shell forging)	0.03	0.71	20.0°F(b)	-30
Lower Shell Forging * Heat # 50D102/50C97-1-1	0.06	0.76	37.0°F(b) 12.9°F(c)	-30
Circumferential Weld * (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	0.03	0.67	41.0 F(b) 25.7F(c)	40
Circumferential Weld * (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	0.04	0.46	54.0°F(b)	-25

* Beltline Region Materials

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.
(b) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev.2, Position 1.1
(c) Chemistry Factor calculated for Cu and Ni values per Regulatory Guide 1.99, Rev. 2, Position 2.1

Table 4.3
Summary of Braidwood Unit 2 Adjusted Reference Temperature (ART's) at 1/4T and 3/4T Location for 14 EFPY[(a)]

Material	14 EFPY	
	1/4T ART (°F)	3/4T ART (°F)
Intermediate Shell Forging Heat # 49D963/49C904-1-1)	3	-8
Lower Shell Forging Heat # 50D102/50C97-1-1	30	11
-Using Surveillance Data	15	11
Circumferential Weld (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	106	85
-Using Surveillance Data	**82** [(a)]	**68** [(a)]
Circumferential Weld (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	29	8
Nozzle Shell Forging Heat # 5P7056	56	46

(a) These ART values were used to calculate the Heatup and Cooldown curves in Figures 2.1 and 2.2 using the 1996 Appendix G Methodology.

Table 4.4

Braidwood Unit 2 Calculation of Adjusted Reference Temperatures (ARTs) at 14 EFPY at the Limiting Reactor Vessel Material Weld Metal WF562 (Based on Surveillance Capsule Data)

Parameter	Values	
Operating Time	14 EFPY	
Location[b]	1/4T ART (°F)	3/4T ART(°F)
Chemistry Factor, CF (°F)	25.7	25.7
Fluence(f), n/cm^2 (E>1.0 Mev))[a]	5.03x10^{18}	1.81x10^{18}
Fluence Factor, FF	0.808	0.546
ΔRT_{NDT}= CFxFF(°F)	20.77[c]	14.04
Initial RT $_{NDT,}$, I(°F)	40	40
Margin, M(°F)	20.77	14.04
ART= I+(CF*FF)+M, °F per RG 1.99, Revision 2	82	68

a) Fluence, f, is the calculated peak clad/base metal interface fluence (E>1.0 Mev) =8 37x10^{18} n/cm^2 at 14 EFPY (Reference 3).
b) The Braidwood Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.
c) Using Regulatory Guide 1.99, Revision 2.

Table 4.5

RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence (10^{19}n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging Heat # 49D963/49C904-1-1	1.96	1.18	20	23.6	23.6	-30	17
Lower Shell Forging Heat # 50D102/50C97-1-1	1.96	1.18	37	43.7	34	-30	48
Lower Shell Forging (Using S/C Data) [d]	1.96	1.18	12.9	15.2	34	-30	19
Nozzle Shell Forging Heat # 5P-7056	0.567	0.841	26	21.9	21.9	30	74
Circumferential Weld (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	1.89	1.17	41.0	48.0	48.0	40	136
Circumferential Weld (Intermediate Shell to Lower Shell) (Using S/C Data)	1.89	1.17	25.7	30.1	28	40	98
Circumferential Weld (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	0.567	0.841	54	45.4	45.4	-25	66

(a) Initial RT_{NDT} values are measured values.

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F).

(c) $\Delta RT_{PTS} = CF * FF$

(d) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater that the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the RT_{PTS} with a full σ_Δ margin term.

Table 4.6

RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence (10^{19}n/cm², E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging Heat # 49D963/49C904-1-1	2.94	1.29	20	25.8	25.8	-30	22
Lower Shell Forging Heat # 50D102/50C97-1-1	2.94	1.29	37	47.7	34	-30	52
Lower Shell Forging (Using S/C Data) [d]	2.94	1.29	12.9	16.6	34	-30	21
Nozzle Shell Forging Heat # 5P-7056	0.849	0.954	26	24.8	24.8	30	80
Circumferential Weld (Intermediate Shell to Lower Shell) Weld Seam WF-562 Heat # 442011	2.83	1.28	41.0	52.9	52.9	40	145
Circumferential Weld (Intermediate Shell to Lower Shell) (Using S/C Data)	2.83	1.28	25.7	32.9	28	40	101
Circumferential Weld (Nozzle Shell to Intermediate Shell) Weld Seam WF-645 Heat # H4498	0.849	0.954	54	51.5	51.5	-25	78

(a) Initial RT_{NDT} values are measured values .

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

(d) Surveillance data is considered not credible. In addition the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Reg. Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor then the Postion 2.1 chemistry factor will be used to determine the RT_{PTS} with a full σ_Δ margin term.

5.0 References

1. WCAP-14040-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

2. Letter from G. F. Dick, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood, Units 1 and 2," dated January 16, 1998.

3. WCAP-15626, "Braidwood Unit 2 12 and 14 EFPY Heatup and Cooldown Limit Curves for Normal Operation using Uprated Fluences," January 2001.

4. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY."

5. Braidwood Station Design Change Package 9900519 (Setpoint Scaling Change Request 00-106), "Revise Unit 2 Low Temperature Overpressure Protection System setpoints/Scaling for Pressurizer Power Operated relief Valves."

6. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.

7. WCAP-11188, "Commonwealth Edison Company, Braidwood Station Unit 2 Reactor Vessel Surveillance Program," December 1986.

8. WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 2000.

9. WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", T.J. Laubham, September 2000.

Attachment 4

List of Changes to the Unit 2 PTLR

Attachment 4
List of Changes to the Braidwood Unit 2 PTLR

Page	Section	Previous	Revised to	Basis
1	1.0	This PTLR for Unit 2 has been prepared in accordance with the requirements of TS 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.	This Pressure and Temperature Limits Report (PTLR) for Braidwood Unit 2 has been prepared in accordance with the requirements of Braidwood Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)". Revisions to the PTLR shall be provided to the NRC after issuance.	Editorial change. Sentence reworded for clarification.
1	2.0	Reference 16	Revised to Reference 2. Reference numbers throughout the Unit 2 PTLR have be changed to reflect either changes in the actual reference or changes in the numbering of existing references to have them appear in numerical sequence through the document.	The basis for actual reference changes will be addressed in the Reference section of this list. The renumbering is an editorial change.
2	2.3	The required enable temperature for the PORVs shall be [greater than or equal to] 350 degrees F RCS temperature.	Deleted. The statement is confusing.	The remaining statements on the Braidwood Procedures for arming and disarming the LTOP are sufficient.
10	2.2	The note wording contains the term "maximum allowable list setpoints"	The wording of the note now states: "determine nominal lift setpoints "	The wording change is made to be consistent Section 2.2 and with setpoint terminology.
11	3.0	"The third and final reactor vessel material irradiation surveillance specimens (Capsule W) have been removed and analyzed to determine changes in material properties."	A sentence has been added at the end of the paragraph: "The surveillance capsule testing has been completed for the original operating period."	Provides consistency with the Braidwood Unit 1 PTLR Section 3.0.
14	Table 4.1	The CF formula listed in the middle of the Table has the sum of the fluence factor squared as 6.218.	The value of the fluence factor squared has been corrected to 6.328.	Identified during NRC review. The value of 6.218 listed in the formula expression is incorrect, it is changed to 6.328; however, the actual CF calculation uses the correct value of 6.328 so the CF value of 12.9 F remains correct.
14	Table 4.1	The FF value for Braidwood 1 Capsule X is listed as 1.060 and the FF^2 value listed for the capsule is listed as 1.266.	The FF value for Braidwood Unit 1, Capsule X is 1.060 and the FF^2 value has been corrected to 1.124.	Identified during NRC review. The summation of the FF^2 values uses the correct value so the CF calculation is correct.

Attachment 4
List of Changes to the Braidwood Unit 2 PTLR

Page	Section	Previous	Revised to	Basis
15	Table 4.2	Clarifications and additions to Table 4.2 listings	1) For Closure Head Flange, added "Serial # 2031-V-1 2) For Vessel Flange, added "Heat #" 3) For Nozzle Shell Forging, added "Heat #" 4) For intermediate Shell Forging added "Heat #" and added note - "(also referred to as the Upper Shell forging)" to provide cross reference to NRC RVID. 4) For Lower Shell Forging, added "Heat #" Added Note "Beltline Region Materials" to apply to Nozzle Shell Forging, Intermediate Shell Forging, Lower Shell Forging, and the two welds.	Editorial, to provide cross reference to NRC RVID Editorial. Editorial. Editorial. Editorial. Clarification of materials of interest, WCAP-15368, Rev 0

Attachment 4
List of Changes to the Braidwood Unit 2 PTLR

Page	Section	Previous	Revised to	Basis
20	5.0	1. WCAP-14040-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.	Maintained as Reference 1	The retained references were renumbered to be sequential in the PTLR. Previous References 2, 3, 4, 5, 7, 9, 12, and 13 were deleted. These references were either not used in support of the Unit 2 PTLR or were not required to be specifically identified as a reference, (e.g. Ref.10 is specifically identified within the PTLR).
		2. WCAP 14824, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration for Byron & Braidwood", Revision 2, November 1997 and Errata Sheets (Westinghouse Letter CAE-97-220, CCE-97-304, CAE-97-233 and CCE-97-316).	Deleted, replaced by: 2. Letter from G. F. Dick, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood, Units 1 and 2," dated January 16, 1998	Old Reference 2 was not used.
		3. WCAP-14228, "Analysis of Capsule X from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 1995, Peter, P.A., et al.	Deleted, replaced by: 3. WCAP-15626, "Braidwood Unit 2 12 and 14 EFPY Heatup and Cooldown Limit Curves for Normal Operation using Uprated Fluences," January 2001.	Old Reference 3 was not used.
		4. WCAP-12845, "Analysis of Capsule U from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 1991, Terek, E., et al.	Deleted, replaced by: 4. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY"	Old Reference 4 was not used.

Attachment 4
List of Changes to the Braidwood Unit 2 PTLR

Page	Section	Previous	Revised to	Basis
		5. Westinghouse Letter to Commonwealth Edison Company, CCE-96-104, "Braidwood Unit 2 LTOPS Setpoints Based on 16 EFPY P/T Limits," January 24, 1996.	Deleted, replaced by: 5. Braidwood Station Design Change Package 9900519 (Setpoint Scaling Change Request 00-106), "Revise Unit 2 Low Temperature Overpressure Protection System setpoints/Scaling for Pressurizer Power Operated relief Valves."	Old Reference 5 was not used. Revised Reference 5 was added. It provides the basis for the revised Unit 2 LTOP Setpoints.
		6. WCAP-11188, "Commonwealth Edison Company, Braidwood Station Unit 2 Reactor Vessel Surveillance Program," December 1986.	Became Reference 7, Reference 6 is now: 6. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.	Editorial.
		7. WCAP-14970, "Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation", T. J. Laubham, October 1997 and Errata Sheets (Westinghouse Letter CAE-97-210, CCE-97-289, CAE-97-232 and CCE-97-315)?.	Deleted, replaced by: 7. WCAP-11188, "Commonwealth Edison Company, Braidwood Station Unit 2 Reactor Vessel Surveillance Program," December 1986.	Old Reference 7 was not used.
		8. WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", T.J. Laubham, September 2000.	Became Reference 9, Reference 8 is now: 8. WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 2000	Editorial.
		9. 10CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," (PTS Rule), January 18, 1996.	Deleted, replaced by: 9. WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", T.J. Laubham, September 2000.	Old Reference 9 was not required.

Attachment 4
List of Changes to the Braidwood Unit 2 PTLR

Page	Section	Previous	Revised to	Basis
		10. U.S. Nuclear Regulatory Commission Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.	Deleted.	Old Reference 2 was not required.
		11. WCAP-15626, "Braidwood Unit 2 12 and 14 EFPY Heatup and Cooldown Limit Curves for Normal Operation using Uprated Fluences," January 2001.	Became Reference 3.	Editorial.
		12. Exelon Document ID # DG01-000125, "Power Uprate-Unit 2 LTOPS," R.D. Koening, dated February 20, 2001.	Deleted.	Old Reference 12 was not used.
		13. Westinghouse Letter CAE-01-053/CCE-01-053, "Transmittal of LTOPs Final Report for Braidwood Unit 2", April 25, 2001.	Deleted.	Old Reference 13 was not used.
		14. Westinghouse Calculation CN-EMT-01-8, "Braidwood Units 1 and 2, Development of New Pressure Temperature Limit Curves and Evaluation of Byron Units 1 and 2 PT Curves EFPY"	Became Reference 4.	Editorial.
		15. NRC Letter from R. A. Capra to O.D. Kingsley, Commonwealth Edison Company, "Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2, Acceptance for referring of pressure temperature limits report, (M98799, M98800, M98801, and M98802)," January 21 1998.	Became Reference 6.	Editorial.
		16. WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program," March 2000.	Became Reference 8	Editorial